

January 13, 1998

Mr. E. E. Fitzpatrick
Executive Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107-1395

SUBJECT: NOTICE OF VIOLATION AND NRC INSPECTION REPORT
NO. 50-315/97018(DRP); 50-316/97018(DRP)

Dear Mr. Fitzpatrick:

On November 7, 1997, the NRC completed an inspection at your D. C. Cook 1 and 2 reactor facilities. The enclosed report presents the results of that inspection.

During the 6-week period covered by this inspection report, the inspectors observed that the plant was operated in a safe manner, maintenance was generally performed well, and radiological work practices were properly followed.

Based on the results of this inspection, one violation of NRC requirements was identified. The violation is cited in the enclosed Notice of Violation (Notice), and the circumstances surrounding it are described in detail in the subject inspection report. The violation is of concern because the inspectors identified two examples of inadequate operations procedures, that were used for cross-tying safety-related buses, containing insufficient guidance to ensure that the activities were properly conducted. A third example of the violation identified a procedure that contained incorrect set points and is of concern because the emergency operating procedure for responding to reactor trips and safety injections contained values different than the plant set-point document.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

E. Fitzpatrick

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In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response will be placed in the NRC Public Document Room.

Sincerely,

/s/ Marc L. Dapas

Geoffrey E. Grant, Director
Division of Reactor Projects

Docket No. 50-315, 50-316
License No. DPR-58, DPR-74

Enclosures: 1. Notice of Violation
 2. Inspection Report
 No. 50-315/970018(DRP); 50-316/97018(DRP)

cc w/encls: A. A. Blind, Site Vice President
 John Sampson, Plant Manager
 Richard Whale, Michigan Public
 Service Commission
 Michigan Department of
 Environmental Quality

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response will be placed in the NRC Public Document Room (PDR).

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CONCURRENCES

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NOTICE OF VIOLATION

Indiana Michigan Power Company
Donald C. Cook Nuclear Power Plant

Docket No. 50-315; 5-316
License No. DPR-58; DPR-74

During an NRC inspection conducted from September 26 through November 7, 1997, one violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

1. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality be prescribed by procedures of a type appropriate to the circumstances and be accomplished in accordance with these procedures.

Contrary to the above,

- A. On October 17, 1997, the inspectors identified that procedure 02-OHP 4021.082.003, Revision 3, "Feeding 600 Volt Buses Through Bus Tie Breakers," was not appropriate to the circumstances in that it allowed, under certain circumstances, an excessive load to be placed on the emergency diesel generators.
- B. On October 17, 1997, the inspectors identified that procedure 02-OHP 4021.082.013, Revision 2, "Isolating, Transferring and Restoring A 250 VDC Load," was not appropriate to the circumstances in that it failed to contain adequate guidance to ensure that battery cross-ties would not be overloaded.
- C. On October 22, 1997, the inspectors identified that Emergency Operating Procedure E-O, 01 [02] - OHP 4023.E-0, Revision 14 [12], "Reactor Trip or Safety Injection," was not appropriate to the circumstances in that 12 examples were identified where the set point for a reactor trip or safety injection as stated in E-O was not as stated in the plant set point document.

This is a Severity Level IV violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, Indiana Michigan Power Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previously docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portion that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated at Lisle, Illinois
this 13th day of January 1998

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No.: 50-315, 50-316

License No.: DPR-58, DPR-74

Report No.: 50-315/97018(DRP); 50-316/97018(DRP)

Licensee: Indiana and Michigan Power
500 Circle Drive
Buchanan, MI 49107-1395

Facility: Donald C. Cook Nuclear Generating Plant

Location: 1 Cook Place
Bridgman, MI 49106

Dates: September 26, 1997, through November 7, 1997

Inspectors: B. L. Bartlett, Senior Resident Inspector
B. J. Fuller, Resident Inspector
J. D. Maynen, Resident Inspector
E. R. Schweibinz, Project Engineer

Approved by: Bruce L. Burgess, Chief
Reactor Projects Branch 6

EXECUTIVE SUMMARY

D. C. Cook Units 1 and 2 NRC Inspection Report No. 50-315/97018(DRP); 50-316/97018(DRP)

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 6 week period of resident inspection and includes the followup to issues identified during previous inspection reports.

Operations

- The licensee identified that during a 4 day period six human performance errors by licensed and non-licensed operators occurred. None of the errors resulted in personnel injuries, equipment damage or an engineered safety features actuation. A Non-cited violation for a failure to follow procedures was issued (Section O1.2).
- The inspectors concluded that the licensee's initial plans and procedures to cross-tie safety-related electrical buses lacked adequate analysis and controls to support plant operation in the proposed configuration. Two examples of a violation for procedural inadequacy were identified. The inspectors were concerned that the licensee did not conduct an adequate evaluation of cross-tying 250 Vdc buses until questioned by the inspectors (Section O3.1).
- The inspectors identified a discrepancy between the pressurizer pressure low safety injection set point as referenced in emergency operating procedure E-0 and as listed on a control board operator aid. While evaluating the inspectors' questions concerning this discrepancy, the licensee identified discrepancies between the plant set-point document and reactor trip set points as listed E-0. The inaccurate procedure was a third example of a violation of NRC requirements (Section O3.2).

Maintenance

- The 2 AB D/G experienced a number of electrical and mechanical failures since May 1997. Two valid run failures resulted in the 2 AB D/G being placed on an accelerated testing frequency. The inspectors were concerned that these failures were indicative of poor material condition. An inspection followup item was opened to track resolution of the material condition of the 2 AB D/G (Section M2.1).
- Following a failure of the 2 AB D/G flywheel end exhaust manifold bracket, the licensee discovered that required jam nuts on the bracket bolts were missing from two emergency diesel generators, 1CD D/G and 2 AB D/G. The licensee speculated that the missing jam nuts may have allowed the bracket bolt to come loose, resulting in a fatigue failure of the bracket; however, the minor modification package paperwork indicated that the jam nuts had been installed. An unresolved item was opened pending a review of the licensee's investigation into the root cause of the bracket failure (Section M2.2).
- The control air system safety valves appeared to be properly installed and dedicated as safety grade components. The inspectors questioned the use of work procedures annotated for non-safety-related work to install safety-related valves; however, no violations of NRC requirements were identified (Section M3.1).

- The inspectors identified unsecured foreign material near the recirculation sump in the Unit 2 lower containment. The sump was not required by Technical Specifications to be operable, and the amount of material would not have significantly degraded the performance of the sump. This was a violation of minor significance (Section M4.1).

Engineering

- Engineering personnel were involved in several of the issues discussed in this report (refer to Section O3.1, Procedures for Cross-Tying 250 Vdc Buses During Maintenance Activities (Unit 2), and Section O3.2, Emergency Operating Procedures Containing Incorrect Set points). Engineering support to the rest of the licensee organization appeared to be good, but as noted in Section O3.1, the support was supplied in response to NRC questions (Section E1).

Plant Support

- The inspectors identified lights under a temporary trailer that were inoperable. The specific root cause was not identified (Section S2).

Report Details

Summary of Plant Status

Unit 1 remained in Mode 5, Cold Shutdown, during this inspection period. The unplanned outage was in response to NRC and licensee concerns with the operability of the containment recirculation sump and other engineering issues.

Unit 2 was in Mode 5, Cold Shutdown, at the beginning of this inspection period. The Unit was in an unplanned outage that was in response to NRC and licensee concerns with the operability of the containment recirculation sump and other engineering issues. On October 26, 1997, the Unit entered a refueling outage that had been scheduled to start September 26, 1997. At the end of the inspection period the Unit was in Mode 6, Refueling.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707 , 60710, and 86700)

Using the referenced inspection procedures, the inspectors conducted frequent reviews of ongoing plant operations. The conduct of operational activities that were observed was generally good. Specific events and noteworthy observations are detailed in the sections below. The inspectors noted that command and control during refueling activities appeared to be excellent.

O1.2 Personnel Errors While Shut Down (Both Units)

a. Inspection Scope (71707)

The inspectors performed follow up inspection for a series of personnel errors that occurred between the dates of October 24 and October 28, 1997. All of the personnel errors were either self-revealing or identified by licensee personnel. Documentation reviewed included:

- Condition Report (CR) 97- 2972, Overfill of reactor cavity and introduction of 1,900 gallons of water into Unit 2 lower containment
- CR 97-2973, The wrong type of oil was added to the Woodward Governor of the Unit 1 CD Diesel Generator (D/G)
- CR 97-2995, The fuel transfer cart was sent back to Unit 2 containment without the fuel assembly being unloaded
- CR 97-3027, Water sprayed from Unit 2, Reactor Coolant Pump 21 and 24 Number 2 seals
- CR 97-3044, The wrong valve was verified closed resulting in a small portion of the Unit 2 Refueling Water Storage Tank (RWST) pipe tunnel being contaminated
- CR 97-3045, Four hundred gallons of primary water were inadvertently added to

the South Boric Acid Storage Tank (BAST)

- Operations Standing Order (OSO) .131, Revision 0, Clearance Restoration

b. Observations and Findings

During a 4 day period from October 24 to October 28, 1997, the operations department experienced a number of personnel errors. The six errors were documented in the CRs listed above. None of the errors resulted in personnel injuries or equipment damage.

Licensee management and NRC inspector review determined that each of the errors had separate human failure causes. Five of the errors referenced in the CRs above involved a failure to follow procedure and the sixth error (CR 97-3027) involved an inadequate procedure. These errors resulted from the actions of both licensed and non-licensed operators.

Licensee management instituted prompt corrective action consisting of the following:

- A 1 hour human performance timeout was held for the crew involved in the first two errors listed above. The timeout focused on the causes for the errors and the prevention of future errors.
- The operators involved wrote lessons learned memoranda and sent them to other operations department personnel.
- The shift managers discussed the importance of error free performance prior to the start of each shift for several shifts following the events described above.
- The plant manager, operations superintendent, and shift manager held discussions with various operators to stress the importance of not getting into a rush, error free job performance, the need to communicate, the importance of pre-job briefs, and the need to pay attention to the job at hand.
- A human performance timeout was also held with the plant work schedulers and the outage management team. This timeout focused on the need to control production pressures and perform in a methodical manner.
- OSO.131 was issued to ensure that personnel verified all vents, drains and inter-system connections listed on a specific clearance were independently verified in the restored position prior to introducing fluid to the system. Previously, an operator would perform the valve lineup and then introduce fluid to the system without verification. The previous method lead directly to the localized contamination of the RWST pipe tunnel discussed in CR 97-3044.

By the end of this inspection period approximately 20 error free operating shifts had occurred. Increased management attention by the first line supervisors and above was evident and contributed to the error free operation.

Problems with procedural adherence were noted in Inspection Reports No. 50-315/95009(DRP) and 50-315/95010(DRP). Since then, the licensee had made steady improvement in the operators' procedural adherence.

Technical Specification 6.8.1 required, in part, that written procedures shall be established, implemented and maintained covering the applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Rev 2, February 1978. Regulatory Guide 1.33, Appendix A, listed typical safety-related activities which should be covered by written procedures.

These six examples of licensee-identified and corrected violations of Technical Specification (TS) 6.8.1 for failure to follow and provide adequate procedures are being treated as a Non-Cited Violation (NCV 50-315/97018-04 and NCV 50-316/97018-04) consistent with Section VII.B.1 of the NRC Enforcement Policy.

c. Conclusions

During a 4 day period, six human performance errors by licensed and non-licensed operators occurred. The licensee identified all six errors, informed the resident inspectors, and took prompt and extensive corrective actions. A non-cited violation for a failure to follow procedures was identified.

O3 Operations Procedures and Documentation

O3.1 Procedures for Cross-Tying 250 Vdc Buses During Maintenance Activities (Unit 2)

a. Inspection Scope (71707 and 62707)

During a routine review of the licensee's plans to replace the CD battery, the inspectors questioned the licensee's plan to cross-tie certain components. Specifically, the inspectors questioned the licensee's plan to maintain both trains of 250 Vdc equipment operable while cross-tying the two safety-related trains. Procedures and documentation reviewed included:

- 02-Operations Head Procedure (OHP) 4021.082.013, Revision 2, Revision 1, and Revision 0, "Isolating, Transferring and Restoring A 250 VDC Battery Load"
- Plant Managers Procedure (PMP) - 4100, Revision 5, "Plant Shutdown Safety and Risk Management"
- 02-OHP 4021.082.001, Revision 0, "4kV Buses Power Source Transfer and De-energizing and Re-energizing a Safeguard Bus"
- 02-OHP 4021.082.003, Revision 3, "Feeding 600 Volt Buses Through Bus Tie Breakers"
- Updated Final Safety Analysis Report (UFSAR), Section 8.1, "Electrical Systems Design Bases"
- UFSAR Section 8.3.4, "250 Volt DC System"

- UFSAR Section 14.2.1, "Fuel Handling Accident Analysis"
- UFSAR Section 14.1.1, "Uncontrolled Rod Withdrawal"
- UFSAR Questions 8.6 and 8.18 concerning the cross-tying of 250 Vdc electrical buses
- TS 3.8.2.4, and bases, "D.C. Distribution - Shutdown"
- Licensee Technical Specification Clarification Number 12, Revision 1 (Canceled), "D. C. Distribution - Operating"
- Memo from L. P. DeMarco to R. K. Gillespie, dated October 13, 1997, "Battery 2 AB /2 CD Crosstie Operation"
- Memo from G. P. Arent to T. P. Beilman, dated October 17, 1997, "Operability Considerations Related to CD Battery Loads While Cross-tied to the AB Battery and Charger"
- Memo from G. P. Arent to R. O. Heathcote, dated October 18, 1997, "D. C. Cook Unit 2, Safety Review Memorandum for Change Sheet No. 1 to 02-OHP 4021.082.013, Isolating, Transferring and Restoring A 250-volt D.C. Battery Load"
- Memo from G. P. Arent to T. P. Beilman, dated October 23, 1997, "Issues Associated with Cross-Tie Operation of the Unit 2 AB and CD Batteries"
- Memo from J. R. Sankey to File, dated October 30, 1997, "Verification of Electrical Distribution System Cross-Tie Capabilities Have Been Properly Evaluated For Use In Procedures"
- CR 97-3068, The operations procedure for cross-tying 600 Vac buses gives erroneous limits for maximum loads

b. Observations and Findings

The inspectors reviewed the licensee's plan for cross-tying the AB ("B" Train) 250 Vdc bus and the CD ("A" Train) 250 Vdc bus. The licensee planned to cross-tie the buses to perform the scheduled 18-month surveillances on the AB battery and leave the buses cross-tied for the CD battery replacement. The licensee explained that the buses would be cross-tied to make an uninterruptable power source (the 2 AB battery) available to power both buses in the event of a loss of alternating current (ac) power to the battery chargers. This would prevent a loss of breaker control power and 120 Vac instrument power.

Technical Issues

The inspectors determined that the licensee's original plan to cross-tie the 250 Vdc buses failed to adequately address several technical issues. The following issues were

discussed with members of the licensee's scheduling, engineering, operations, and

licensing staff:

- A common mode failure, through the cross-connected CD and AB trains of 250 Vdc power, could cause all 250 Vdc power to be lost. Therefore, both 250 Vdc trains would be technically inoperable while cross-tied, making all breaker control power and 120 Vac instrument power simultaneously inoperable.
- Technical Specification 3.8.1.2, which required that at least one D/G be operable in Modes 5 and 6, had not been adequately considered. The operability of the D/G must be demonstrated, in part, by the capability to start from standby conditions and achieve rated voltage and frequency in less than or equal to 10 seconds. The licensee's original outage schedule would have counted upon the 2 CD D/G being operable while depending upon the cross-tied 250 Vdc D/G control power.
- The operations department procedure governing the transfer of 250 Vdc battery loads, an activity not described in the Updated Final Safety Analysis Report (UFSAR), did not have a 10 CFR 50.59 Safety Evaluation (SE). No SE existed for the original procedure and all subsequent SEs focused only upon the changes to the procedure. Thus the process of tying the safety-related buses together, which was the purpose of the procedure, did not have an SE which determined that the plant would not be operated outside the design as described in the UFSAR.

Adequacy of 10 CFR Part 50.59s was one of the principle concerns of a recent NRC Architect Engineering (AE) Team inspection (50-315/97-201). As this issue is identical to issues the AE Team identified regarding some 50.59 reviews, this item will be incorporated into the NRC's regulatory response to the AE Team findings. In the interim, it will be tracked as an Unresolved Item (50-316/97-018-03(DRP)).

- Licensee procedure PMP-4100, addressing shutdown safety and risk management, required the scheduling department to develop the outage schedule with consideration of risk to the reactor. The inspectors determined that the outage schedule did not take into account the increased operability runs of the 2 AB D/G, however, the schedule had not yet been officially approved at the time of the inspectors' review. The schedule had been sent to the Plant Nuclear Safety Review Committee (PNSRC) and PNSRC members had questions concerning the 2 CD battery replacement. The PNSRC members were concerned about making the 2 AB D/G inoperable in accordance with the surveillance test procedure. The 2 AB D/G surveillance test, performed while the CD battery replacement rendered the 2 CD D/G inoperable, would make both D/Gs simultaneously inoperable.

Pending the response to questions concerning the operability of the 2 CD D/G while dependent upon the opposite trains batteries, the schedule had been approved with a hold to resolve the CD battery replacement issue.

After the inspectors discussed these issues with the licensee, the licensee's scheduling, licensing, and engineering staff conducted an evaluation of cross-tying the 250 Vdc,

480 Vac, and 600 Vac buses. The licensee's reviews identified several additional issues:

- As required by TS, both residual heat removal (RHR) loops must be operable at all times when the reactor coolant system is not filled and vented, the refueling cavity is less than 23 feet, and fuel is in the reactor vessel. The operability of the D/Gs and the RHR pump's attendant electrical supplies had not been adequately considered.
- Two of the three channels from each train of the radiation monitoring system (RMS) were required to be operable while performing a containment purge during Mode 6. An automatic containment ventilation isolation function from the RMS channels to the purge containment isolation valves was also required. With the 250 Vdc buses cross-tied, the automatic isolation function would lack independent power supplies and would therefore be inoperable.
- Procedure 02-OHP 4021.082.003, which could be performed in any operating mode, erroneously allowed a load of up to 560 amps on the 4 kV side of the engineered safeguards system transformer. No engineering calculation was found to support this allowable loading. An engineering calculation (PS-600VD-012) was performed on October 30, 1997, and determined that the maximum loading was 225 amps. If the 560 amp limitation had been maintained, the D/G would have been overloaded by about 40 kW or the cross tie load limit would have been exceeded.

The failure of procedure 02-OHP 4021.082.003, to provide appropriate instructions was an example of a violation of 10 CFR Part 50, Appendix B, Criterion V, which requires that activities affecting quality be prescribed by procedures appropriate to the circumstances and be accomplished in accordance with these procedures (50-316/97018-01a(DRP)).

Procedure 02-OHP 4021.082.013, Revision 2, "Isolating, Transferring and Restoring A 250 Vdc Battery Load," did not provide limits for controlling dc bus loading on cross-tied busses. The failure of procedure 02-OHP 4021.082.013 to provide appropriate instructions for cross-tying the 250 Vdc busses was an example of a violation of 10 CFR Part 50, Appendix B, Criterion V (50-316/97018-01b(DRP)).

Resolution of the Technical Issues

The licensee determined that cross-tying the buses would be acceptable provided that proper controls were in place to control bus loading, and a change sheet was issued which placed administrative controls on non-essential 250 Vdc loads.

Additionally, the licensee verified that fuse coordination and cable capacity were adequate to support cross-tie operation. These controls would ensure that the bus, which was connected to its battery, would not be lost due to a fault occurring on the opposite train 250 Vdc bus. Therefore, the licensee concluded that the bus which was connected to its associated battery would be operable in accordance with Technical Specification 3.8.2.4, and the opposite train bus would be "available" per PMP-4100.

The outage schedule was changed to prevent cross-tying the 250 Vdc buses prior to filling the refueling cavity to greater than 23 feet if there was fuel in the reactor vessel. Technical

Specifications allowed one train of RHR to be inoperable if there were greater than 23 feet of water over the reactor vessel.

The CD battery replacement was initially scheduled for the period when the reactor was defueled; thus, the 2 AB D/G could be run when the 2CD D/G was not operable due to the CD battery being disconnected from the 2 CD 250 Vdc bus. However, the licensee determined that while in Modes 5 and 6, the TSs allowed the testing of the 2 AB D/G without it being loaded onto the grid. This in turn ensured that the D/G would not be declared inoperable during the test.

An administrative control was implemented which prohibited the operation of containment purge while core alterations were in progress with either 250 volt battery disconnected from its associated bus. This action ensured that the containment purge isolation valves were in their safety-related positions while independent power supplies were not available.

The licensee's operations staff planned to develop a procedure for manually operating the second train of emergency core cooling system breakers in the event of a loss of dc control power to these breakers. The licensee concluded that adequate time for manual actions exist to ensure that the RHR system could meet its specified function in Modes 5 or 6.

Inspector Verification of the Licensee's Conclusions

The inspectors independently verified the basis for the licensee's conclusions and determined that the licensee's administrative controls appeared adequate to ensure compliance with TSs. The inspectors were concerned that these controls were not implemented until after the inspectors questioned the practice of cross-tying safety-related buses. In addition, the UFSAR did not specifically address the use of the cross-tie breakers for normal conditions such as maintenance.

There was good engineering support to the operations department regarding the procedure for cross-tying buses. This support occurred mainly after the inspectors questioned the licensee's practice of cross-tying buses.

c. Conclusions

The inspectors concluded that the licensee's initial plans and procedures to cross-tie safety-related electrical buses lacked adequate analysis and controls to support plant operation in the proposed configuration. Two examples of a violation for procedural inadequacy were identified. The inspectors were concerned that the licensee did not conduct an adequate evaluation of cross-tying 250 Vdc buses until questioned by the inspectors.

O3.2 Emergency Operating Procedure Containing Incorrect Set points (Both Units)

a. Inspection Scope (71707)

During a routine tour of the Unit 2 control room, the inspectors identified a discrepancy between the pressurizer pressure low, safety injection set point as stated in a plant procedure and on a control board operator aid. Procedures and documentation reviewed included:

- 02-OHP 4023.E-0, Revision 12, "Reactor Trip or Safety Injection" (E-0)
- 01-OHP 4023.E-0, Revision 14, "Reactor Trip or Safety Injection" (E-0)
- CR 97-2591, The values on the control board in the control room and Emergency Operating Procedures (EOP) values for pressurizer low pressure do not reflect the current value listed in the Plant Set Point Document (PSPD) or TS.

b. Observations and Findings

On October 22, 1997, the inspectors observed that the safety injection (SI) set point for low pressurizer pressure as stated on a control board operator aid was 1,908 pounds per square inch gage (psig). The safety injection set point for low pressurizer pressure as stated in procedure E-0 was 1,900 psig.

The inspectors were told by the Unit Supervisor that set points were put in procedure E-0 in a manner which would make them easy to verify. This answer did not seem correct as the set point for the pressurizer pressure low reactor trip was 1,966 psig. Thus the 1,900 psig number was easy to quickly check on the control board meter but 1,966 psig would not be easy to quickly check.

Licensee personnel performed a review of the set points for various reactor trip and safety injection parameters and compared them to the EOPs and TS. Licensee personnel identified that of the twenty-one reactor trip set points and four safety injection set points for each unit, the following discrepancies existed (erroneous numbers in **bold**).

Unit 2 Safety Injection Set points

- Pressurizer pressure low :
Control Board Aid read **1908** psig versus a set-point and TS limit of 1900 psig

Unit 1 Reactor Trip Set points

- Power Range (PR) Negative Rate Trip:
EOP listed **5 percent** decrease in 2 seconds versus a set point of 4.5 percent decrease in 2 seconds and a TS limit of ≤ 5 percent in ≥ 2 seconds
- PR Positive Rate Trip:
EOP listed **5 percent** increase in 2 seconds versus a set point of 4.5 percent increase in 2 seconds and a TS limit of ≤ 5 percent in ≥ 2 seconds

- Pzr Pressure High:
EOP and Control Board Aid listed **2,378 psig** versus a set point and TS limit of 2,385 psig
- Pzr Level High:
EOP listed **91 percent** versus a set point and TS limit of 92 percent
- Low Feedwater Flow < Steam Flow with Low Steam Generator (S/G) Level:
EOP listed Feedwater Flow < Steam Flow by 710,000 pounds mass per hour (lbm/hr) with S/G level **26 percent** versus a set point and TS limit of Feedwater Flow < Steam Flow by 710,000 lbm/hr with S/G level 25 percent

Unit 2 Reactor Trip Set Points

- PR Negative Rate Trip:
EOP listed **5 percent** decrease in 2 seconds versus a set point of 4.5 percent decrease in 2 seconds and a TS limit of ≤ 5 percent in ≥ 2 seconds
- PR Positive Rate Trip:
EOP listed **5 percent** increase in 2 seconds versus a set point of 4.5 percent increase in 2 seconds and a TS limit of ≤ 5 percent in ≥ 2 seconds
- Pzr Pressure High:
EOP and Control Board Aid listed **2,378 psig** versus a set point and TS limit of 2,385 psig
- Pzr Pressure Low:
EOP and Control Board Aid listed **1,966 psig** versus a set point and TS limit of 1,950 psig
- Pressurizer Level High:
EOP listed **91 percent** versus a set point and TS limit of 92 percent
- Low Feedwater Flow < Steam Flow with Low S/G Level:
EOP listed Feedwater Flow < Steam Flow by 1,470,000 lbm/hr with S/G level **26 percent** versus a set point and TS limit of Feedwater Flow < Steam Flow by 1,470,000 lbm/hr with S/G level 25 percent

The control board operator aids that were in error were removed on October 24, 1997. Interviews with the licensed operators determined that most of them were unaware of the errors. Most operators stated that when they needed to look up the set-point for a reactor trip or safety injection they used the E-0 procedure even though both the EOP and the control board operator aids were found to contain errors.

The root causes for the failure to have accurate and up to date set points in the EOP were still being evaluated by licensee personnel at the end of the report period. The inspectors determined that the incorrect value for the Unit 2 low Pressurizer Pressure SI of 1,908 psig had been wrong since 1994.

The failure to have an EOP procedure that referenced the actual reactor trip and/or safety injection set points was another example of a violation of 10 CFR Part 50 Criterion V, which requires that activities affecting quality be prescribed by procedures of a type appropriate to the circumstances and be accomplished in accordance with these procedures (50-315/97018-01c(DRP)). The safety significance of these errors was low as the differences between the actual set points and the values specified in the EOP and/or on the operator aids were small enough that the differences could not be distinguished on the control board meter faces.

c. Conclusions

The inspectors identified a discrepancy between the pressurizer pressure low safety injection set point as referenced in emergency operating procedure E-0 and as listed on a control board operator aid. While evaluating the inspectors' questions concerning this discrepancy, the licensee identified discrepancies between the plant set-point document and reactor trip set points as listed E-0. The inaccurate procedure was a third example of a violation of NRC requirements.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62707 and 61726)

Portions of the following maintenance job orders, action requests, and surveillance activities were observed or reviewed by the inspectors:

- 12 Maintenance Head Procedure (MHP) 4030. Surveillance Test Procedure (STP) .046, Revision 5, "Emergency Diesel Generator System 18 Month Inspection"
- (JO) Job Order C42327, Install Design Change Package (DCP) - 854, control air system relief valves, on Unit 1 control air system
- JO C42352, Install DCP-854, control air system relief valves, on Unit 2 control air system
- 02-Engineering Head Procedure (EHP) 4030.STP.217A, Revision 3, "DG2CD Load Sequencing and Engineered Safety Feature Testing (ESF)"
- 02-EHP 4030.STP.217B, Revision 3, "DG2 AB Load Sequencing and ESF Testing"

b. Observations and Findings

The inspectors noted that, in general, most of the outage work activities were performed in a quality manner with procedures present and in use.

During work activities on the Unit 2 AB D/G, the inspectors questioned the workers

regarding the torque of the engine to foundation nuts. It appeared to the inspectors that a step in the work procedure required the torque to be checked at 1,950 foot-pounds (ft-lbs), but the workers were checking the torque at 2,006 ft-lbs. In response to the inspectors' question, the workers agreed with the inspectors and stopped work to further research the question. Subsequently, the licensee and the inspectors determined that the workers were performing the step properly but that the step was poorly worded. It was also determined that this question had previously been raised by the workers and they were verbally informed of the proper way to perform the step. Maintenance personnel wrote a Condition Report to document that a poorly worded procedure step was not revised when it was originally questioned by the workers. This step and other similarly worded steps were revised to more clearly specify the requirements prior to the workers resuming the maintenance activity.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 AB D/G Poor Material Condition (Unit 2)

a. Inspection Scope (62707)

On October 13, 1997, operations personnel were attempting to run the Unit 2 AB D/G for a weekly surveillance. During the run, a high cylinder exhaust differential temperature was noticed, and the 2 AB D/G surveillance was stopped. An additional problem in the control room circuitry was discovered when attempting to shut down the engine for troubleshooting. The inspectors followed the licensee's troubleshooting effort and reviewed the following documents:

- CR 97-2810, Unacceptable Maintenance Rule performance of the Unit 2 train B emergency diesel generator
- **02-OHP 4030.STP.027AB, Revision 10, "AB Diesel Generator Operability Test (Train B)"
- 02-OHP 4021.032.001AB, Revision 4, "DG2 AB Operation"

b. Observations and Findings

On October 13, 1997, the 2 AB D/G was being run for a weekly surveillance. The 2 AB D/G had been on an increased frequency surveillance testing schedule as required by TS 4.8.1.1.2 due to a previous failed start. During the run, operations personnel noticed that the number 3 rear cylinder exhaust temperature was low and that the other cylinder exhaust temperatures were higher than normal. This resulted in a high cylinder exhaust differential temperature, and the 2 AB D/G was manually stopped. During the engine shutdown, after the 2 AB D/G Stop/Run control switch had been placed in the stop position, the 2 AB D/G unexpectedly returned to rated speed once the switch was returned to the After-Start position. The operator then manually tripped the 2 AB D/G from the control room, stopping the engine.

During troubleshooting, the licensee identified two separate problems. The high pressure fuel line to the number 3 rear cylinder had developed a through-wall leak, and the switch contacts for the 2 AB D/G Start-Stop switch had also failed. Both the fuel line and the

switch were replaced. Additionally, the leaking fuel line had allowed about 75 gallons of fuel oil to contaminate the 2 AB D/G lubricating oil, requiring the licensee to replace the lubricating oil.

The inspectors observed that since May 1997 nine mechanical and electrical failures have been documented on the 2 AB D/G. As a result of exceeding the limit for functional failures, the licensee moved the 2 AB D/G to Maintenance Rule category (a)(1), which required a plan to improve the performance of the equipment. At the end of this inspection report period, the licensee had not yet completed the plan for monitoring the 2 AB D/G. The inspectors considered the issue involving the material condition of the Unit 2 AB D/G an inspector followup item pending a review of the licensee's monitoring plan (50-316/97018-05).

c. Conclusions

The 2 AB D/G experienced a number of electrical and mechanical failures since May 1997. Two valid run failures resulted in the 2 AB D/G being placed on an accelerated testing frequency. The inspectors were concerned that these failures were indicative of poor material condition. An inspection followup item was opened to track resolution of the material condition of the 2 AB D/G.

M2.2 Diesel Generator Exhaust Manifold Brackets (Both Units)

a. Inspection Scope (62707)

On October 19, 1997, while running the 2 AB D/G for an eight hour surveillance test, the flywheel end exhaust manifold bracket failed. The inspectors followed the licensee's evaluation of the event and followup testing. In addition, the inspectors reviewed the following documents:

- 12-Minor Modification (MM) - 438, "Replace the emergency diesel generator exhaust manifold structure supports"
- 12-DCP-861, "Enhancement of the bracket tab of flywheel end support assembly of exhaust manifold for Emergency Diesel Generators"
- PMP-5040 MOD.002, Revision 8, "Minor Modifications"
- PMP-5040 MOD.003, Revision 6, "Plant Modifications"
- **12 Construction Head Procedure (CHP) 5021.MCD.001, Revision 2, "Fabrication and Installation of Safety-related/Safety Interface Component Supports, Hangers, and Restraints"
- **02-OHP 4030.STP.027AB, Revision 10, "AB Diesel Generator Operability Test (Train B)"
- Condition Report 97-2904, During the 8 hour test run for 2 AB diesel generator, the generator end exhaust manifold support bracket broke.

- JO C18428, Replace emergency diesel generator exhaust manifold supports, 1AB D/G
- JO C18424, Replace emergency diesel generator exhaust manifold supports, 1CD D/G
- JO C19480, Replace emergency diesel generator exhaust manifold supports, 2 AB D/G
- JO C19477, Replace emergency diesel generator exhaust manifold supports, 2CD D/G
- Drawing 01-A-EQS-197, Unit 1 AB and CD diesel support arrangement
- Drawing 01-A-EQS-198, Unit 1 AB and CD diesel manifold-exhaust conversion

b. Observations and Findings

During an 8 hour surveillance test on the 2 AB D/G, the flywheel end exhaust manifold bracket failed. The engine was manually shutdown, and the licensee conducted an inspection of the bracket. The failure occurred at the upper bolt hole of the bracket, and examination of the failed bracket revealed evidence that the upper bolt had become loose within the bolt hole. The manifold bracket was installed under MM-438 during the most recent refueling outages for each unit.

The licensee evaluated the possibility of a common mode failure for the D/Gs because MM-438 had been installed on each D/G. The licensee inspected the other D/Gs for indications of failure or wear at the manifold brackets, but no failure or wear was found. The licensee also found that the 1 CD D/G and 2 AB D/G had no jam nuts installed on the flywheel end exhaust manifold bracket; however, the modification drawings indicated that a jam nut was to be installed on each of two bolts on the flywheel end support bracket. The licensee speculated that a missing jam nut may have allowed the bolt to become loose, leading to a fatigue failure of the bracket.

The licensee concluded that the D/Gs would remain operable provided that the jam nuts were installed and the manifold bracket bolts were properly torqued. Jam nuts were installed on the 1 CD D/G, and except for the 2 AB D/G, the bolts on all of the brackets were properly torqued. A new design change, DCP-861, was installed on the 2 AB D/G to repair the flywheel end exhaust manifold bracket. DCP-861 installed a larger bracket which appeared to be more resistant to fatigue failure than the brackets installed under MM-438. The licensee planned to install DCP-861 on the other three diesels during normally scheduled maintenance outages. The inspectors reviewed the licensee's prompt operability determination, corrective actions, and DCP-861 and had no additional concerns.

The inspectors questioned the licensee about the missing jam nuts and reviewed the minor modification package. The MM-438 paperwork indicated that the flywheel end exhaust manifold bracket had been properly installed on all four D/Gs with the jam nuts installed. The inspectors were concerned that the flywheel end exhaust manifold bracket on the 2 AB D/G, a safety-related component, had failed, and that the other three D/Gs had an identical modification installed. Condition report 97-2904 was issued to document the issue, and the

licensee started an investigation into the root cause of the bracket failure. At the end of this report period, the licensee's investigation was not completed; therefore, this issue was considered an unresolved item (50-315/97018-06; 50-316/97018-06) pending the inspectors' review of the licensee's root cause investigation.

c. Conclusions

Following a failure of the 2 AB D/G flywheel end exhaust manifold bracket, two emergency diesel generators, 1CD D/G and 2 AB D/G, were found to be missing required jam nuts on the bracket bolts. The licensee speculated that the missing jam nuts may have allowed the bracket bolt to come loose, resulting in a fatigue failure of the bracket; however, the minor modification package paperwork indicated that the jam nuts had been installed. An unresolved item was opened pending a review of the licensee's investigation into the root cause of the bracket failure.

M3 Maintenance Procedures and Documentation

M3.1 Control Air Header Safety Valve Installation

a. Inspection Scope (62707)

On October 7, 1997, the inspectors observed portions of the control air system safety valve installation, DCP-854, which was being done as part of the NRC Architect Engineer inspection corrective actions. In addition to observing the work at the site, the inspectors reviewed the following documents:

- D. C. Cook (DCC) - Mechanical Engineering (ME) -201- Quality Control Number (QCN), Revision 1, "Material Testing Specifications"
- **12 MHP-5021.001.034, Revision 7, "Safety Valve Bench Testing"
- 12 Material Maintenance Head Procedure (MMP) -3120. Nuclear Engineering Testing Section (NETS) .001, Revision 0, "Receipt Inspection of Safety-related/Interfaced Material and Equipment"
- Dedication Plan Pressure Valve (PV) -1031, Revision 4, "Relief valves to provide over pressure protection for nuclear safety-related air operated components supplied by station air"
- 12-DCP-854, Safety relief valves for control air system and changes to the control air to the auxiliary equipment ventilation system and spent fuel pool ventilation dampers
- JO C42327, Install DCP-854, control air system relief valves, on Unit 1 control air system
- JO C42352, Install DCP-854, control air system relief valves, on Unit 2 control air system
- Condition Report 97-2770, Job order activity work packages for DCP-854 do not

adequately document the unique activity of placing relief valves with safety-related pressure relieving capabilities in non-safety-related control air system

b. Observations and Findings

On October 7, 1997, the inspectors observed part of the safety valve installation on the turbine deck for the Unit 1 control air headers. The job orders and work procedures were at the site and in active use. The inspectors noted good supervisory oversight and proper foreign material exclusion practices.

The control air header safety valves were being installed to protect safety-related air operated valves from experiencing an over pressure condition. As a result, the safety valves were also required to be safety-related, although the control air system is not considered a safety-related system. The inspectors noted that the work packages at the job site were clearly marked "non-safety-related", and the job order for the installation of DCP-854 had no entry regarding whether the job was safety-related or not safety-related.

The inspectors questioned the licensee about the valve installation and reviewed DCP-854. The purpose of the control air system safety valves was to protect down-stream safety-related valves; therefore, each control air system safety valve had a safety-related function. The control air system was not safety-related; therefore, the installation of the valves was not required to be controlled as a safety-related work activity. After discussing the inspectors' questions and reviewing the work packages, the licensee added a material traceability sheet to each package which identified the specific location of each control air safety valve.

The inspectors also reviewed the dedication plan and test results for a sample of the safety valves. No deficiencies were noted, and the valves appeared to be properly dedicated as safety grade material in accordance with the dedication plan.

c. Conclusions

The control air system safety valves appeared to be properly installed and dedicated as safety grade components. The inspectors questioned the use of work procedures annotated for non-safety-related work to install safety-related valves; however, no violations of NRC requirements were identified.

M4 Maintenance Staff Knowledge and Performance

M4.1 Control of Transient Material In Containment (Unit 2)

a. Inspection Scope (71707)

During a routine tour of the Unit 2 lower containment, the inspectors identified unsecured transient material near the recirculation sump. Routine follow up was performed to identify the material's source and the reasons it was unsecured. Procedures and documentation reviewed included:

- Plant Managers Standing Order (PMSO) - 179, Revision 1, "Transient Materials In Containment While The Unit Is Shutdown"
- CR 97-2834, Transient Material Identified Inside Containment By NRC Inspectors
- 12 PMP 2220.001.001, Revision 0, "Foreign Material Exclusion (FME)"
- CR 97-3003, During a walkdown of Unit 2 lower containment (FMEZ) foreign material exclusion zone-2 zone several items were identified
- PMP - 4100, Revision 0, "Plant Shutdown Safety and Risk Management"

b. Observations and Findings

During a routine tour of lower containment on October 15, 1997, the inspectors identified three bags of light weight material. The material was light enough that if water was present, the material could have floated and potentially blocked the recirculation sump. The Unit was in Mode 5 (Cold Shutdown) and a refueling outage was in progress at the time. This material was within 15 feet of the recirculation sump screens. Technical Specifications did not require that the recirculation sump be operable in Mode 5; however, the licensee's shutdown safety procedures did require that the sump be available.

In response to the inspectors findings, licensee personnel performed a tour of lower containment and identified loose plastic spray bottles, spray cans, cloth gloves, empty nylon bags, tie wraps, tape and paper.

During previous refueling outages the licensee has had trouble controlling materials in lower containment. As documented in Inspection Report (IR) 50-315/97004(DRP), Section O1.5:

"During a routine containment tour, the inspectors identified a large accumulation of bagged material in the area directly in front of the grates at the entrance to the ECCS [Emergency Core Cooling System] recirculation sump in lower containment. The materials consisted of scaffolding, S/G [steam generator] eddy current inspection equipment and a large amount of bagged insulation removed in preparation for S/G inspections. The licensee had independently identified that transient material had been allowed to accumulate and was in the process of taking action to remove the material."

In IR 50-315/97004, the licensee had identified the material at the same time as the inspectors and was taking prompt corrective action. However, while the material identified on October 15, 1997, was significantly less than the amount identified during the inspection documented in IR 50-315/97004, this material was identified by the NRC inspectors.

Licensee corrective actions for this new material consisted of:

- Holding a site wide safety timeout on October 17, 1997, and including the control of transient material in lower containment as one of the discussion topics
- Modifying FME procedure 12 PMP 2220.001.001, to add specific requirements for

the control of transient material in containment, including:

- Designating containment below the 617' elevation as FMEZ-2 in Modes 5 and 6 (refueling). This is an intermediate level of FME control and includes the requirement for additional administrative controls and management supervision.
- Ensuring that any unattended material which may be transported by water flow is properly secured.
- Highlighting this issue in several daily plant newsletters
- Having the Radiation Protection (RP) technicians inside containment perform routine tours to identify any unsecured material

During tours, licensee personnel subsequently identified additional material and determined that still further corrective action was necessary. As such, maintenance first line supervisors were tasked with performing periodic tours.

The failure of licensee personnel to follow procedure PMSO-179 for control of FME was a violation of Technical Specification 6.8.1 which required that written procedures shall be established, implemented and maintained covering the applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Rev 2, February 1978. Regulatory Guide 1.33, Appendix A, listed typical safety-related activities which should be covered by written procedures. This failure constituted a violation of minor significance and is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy (NCV 50-316/97018-07).

c. Conclusions

The inspectors identified unsecured foreign material near the recirculation sump in the Unit 2 lower containment. The sump was not required by Technical Specifications to be operable, and the amount of material would not have significantly degraded the performance of the sump. This was a violation of minor significance.

III. Engineering

E1 Conduct of Engineering

During the resident inspection activities, routine observations were conducted in the areas of engineering using Inspection Procedure 37551. During this inspection report period, the licensee's engineering organization expended significant effort on resolving NRC Architect Engineering Team issues. These issues and concerns will be documented in Inspection Report No. 50-315/97-201. Specific follow up inspection activities related to the issues identified in Inspection Report No. 50-315/97-201 will be documented in subsequent inspection reports.

Engineering personnel were also involved in resolving several of the issues discussed previously in this report (refer to Section O3.1, Procedures for Cross-Tying 250 Vdc Buses During Maintenance Activities (Unit 2), and Section O3.2, Emergency Operating

Procedures Containing Incorrect Set points). Engineering support to the rest of the licensee's organization appeared to be good.

IV. Plant Support

P5 Staff Training and Qualification in EP

- P5.1 (Open) Unresolved Item (50-315/97015-02(DRP); 50-316/97015-02(DRP)): Emergency Response Organization Respirator Qualifications (Both Units). During a review of an Operating Experience report, the licensee identified that the program for maintaining operator respirator qualifications did not include a provision for ensuring that the operators maintained corrective lenses available when necessary. The program also failed to include an annual respirator fit testing requirement for members of the Emergency Response Organization who were required to be respirator qualified.

The licensee's short-term corrective actions were to ensure that all personnel who were required to be respirator qualified had completed all of the training and medical requirements for respirator use. At the end of this report period, all members of the Emergency Response Organization have been qualified to use a respirator; however, this unresolved item will remain open pending a review of the licensee's long term corrective actions.

F1 Control of Fire Protection Activities (71750)

During normal resident inspection activities, routine observations were conducted in the area of fire protection activities using Inspection Procedure 71750. No discrepancies were noted.

S2 Status of Security Facilities and Equipment (71750)

During a routine tour of the protected area on October 7, 1997, just prior to sunrise, the inspectors observed that the area underneath a temporary trailer was not lit. The temporary lights underneath the temporary trailer were not operating. The inspectors informed the security shift captain.

The licensee's Modified Amended Security Plan (MASP), Revision 31, required in Section 4.1.3.3 that the protected area be lit. Section 4.1.5 required that personnel perform periodic patrols and that, as a part of the patrols, protected area lighting should be inspected.

The licensee determined that the most probable reason the lights under the trailer were out was because workmen in the area of the RWST had unplugged the lights. Apparently in need of a power cord, they had unplugged the lights to the trailer. Licensee personnel performed a review of the work records of the workers and determined that on the day before this issue was identified, they had stopped work at 5:00 p.m.

The licensee's security staff performed periodic patrols of the protected area and thus missed multiple opportunities to identify this unlighted area. However, it was not determined for certain that the RWST workers were the personnel responsible for unplugging the lights. As other workers were near the area during the night, the precise

time and reason the lights were unplugged could not be determined.

The licensee had periodically measured the lighting level inside the protected area. In response to the inspectors' finding, the licensee changed the procedure to significantly reduce the time between lighting measurements. This was done to provide additional opportunities to identify protected area lighting issues.

While reviewing the MASP, the inspectors identified that Figure 4.1-1, "Protected Area Perimeter Fence and Isolation Zone," failed to show one of the vehicle gates. This gate was identified on other MASP drawings but was not shown on Figure 4.1-1. The licensee was informed and agreed to review and revise as necessary, Figure 4.1-1.

X1 Exit Meeting

The inspectors presented the inspection results to members of the licensee management at the conclusion of the inspection on November 5, 1997. The licensee had additional comments on some of the findings presented.

- Regarding the cross-tying of safety-related electrical buses, the plant manager stated that as a result of the NRC AE Team findings, the licensee's engineering staff had already decided that cross-tie capabilities needed to be further reviewed. The licensee was also evaluating the design basis and how it applied to existing operating procedures. In addition, the plant manager stated that as a result of a recent NRC Information Notice on manual operator action, the licensee's engineering staff would re-evaluate the initial plan to take credit for manual operator action during maintenance activities on the 2 CD Battery.
- Regarding the six human performance errors in the operations department, the operations superintendent stated that even though each item by itself was not significant, operations department management wanted to treat each issue as though it was more significant for the purpose of learning from each of the issues in order to prevent more significant mistakes from occurring. The operations superintendent also stated that operations department personnel had perceived that a production pressure existed. Once management stated that they would take the time to do it right, no matter how long it took, the perceived production pressure was eliminated.

In addition, the operations superintendent stated that 2 years ago the NRC had to identify to D. C. Cook that there were human performance errors. This time, licensee personnel had identified the errors and taken prompt corrective action.

- Regarding the incorrect set points in E-0, the inspector stated that he didn't know if the process error which had occurred affected only E-0, affected other operations department procedures, or if the process error affected all procedures at the plant. The Site Vice-President stated that his staff would find the root causes, determine the extent of the issue, and correct it. The operations superintendent stated that at some time in the past, the licensee had decided not to keep the annunciator response procedures up to date on instrument set points. He further stated that his staff was in the process of updating the annunciator response procedures as appropriate, and the annunciator response procedures would be updated prior to

entry into Mode 2 (reactor critical).

- Regarding the transient material inside Unit 2 containment, the plant manager stated that he was not trying to excuse the as-found condition, but that the amount of material found was relatively small, radiation protection personnel had taken prompt and extensive corrective action, and that overall, performance in this area was significantly improved over that observed in previous outages.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

#K. Baker, Manager, Production Engineering
#P. Barrett, Manager, Performance Assurance
#A. Blind, Site Vice President
#T. Beilman, Scheduling Supervisor
#J. Benes, Supervisor, Balance of Plant Mechanical Systems
#M. Depuydt, Nuclear Licensing
#S. Farlow, Supervisor, I&C Engineering
#M. Finissi, Supervisor, Electrical Systems
#J. Frye, Radiation Protection
#R. Gillespie, Operations Superintendent
#D. Hafer, Manager, Plant Engineering
#J. Kobyra, Manager, Nuclear Engineering
#D. Morey, Chemistry Superintendent
#A. Olvera, Nuclear Licensing
#F. Pisarsky, Supervisor, Mechanical Component Engineering
#T. Quaka, Project Management & Inst. Services
#J. Sampson, Plant Manager
#P. Schoepf, Supervisor, Safety-related Mechanical Systems
#L. VanGinhoven, Supervisor, Materials Management
#A. Verteramo, Nuclear Engineering
#T. Wagoner, Maintenance

#Denotes those present at the November 5, 1997, exit meeting.

INSPECTION PROCEDURES USED

IP 37551	On-site Engineering
IP 60710	Refueling Outage
IP 61726	Surveillance Observations
IP 62703	Maintenance Observation
IP 71707	Plant Operations
IP 71750	Plant Support Activities
IP 86700	Spent Fuel Pool Activities

ITEMS OPENED and CLOSED and DISCUSSED

ITEMS OPENED

50-316/97018-01a	NOV	Failure to have instructions of a type appropriate to the circumstances
50-316/97018-01b	NOV	Failure to have instructions of a type appropriate to the circumstances
50-315/97018-01c 50-316/97018-01c	NOV	Inaccurate and out of date EOPs
50-316/97018-03	URI	Adequacy of operations procedure SE
50-315/97018-04 50-316/97018-04	NCV	Failure to follow procedures
50-316/97018-05	IFI	Material condition of Unit 2 AB emergency diesel generator
50-315/97018-06	URI	Missing jam nuts on Unit 1 CD emergency diesel generator
50-316/97018-07	NCV	FME in Unit 2 containment

ITEMS CLOSED

None

ITEMS DISCUSSED

50-315/97015-02 50-316/97015-02	URI	Not all members of the Emergency Response Organization maintained current respirator qualifications
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LIST OF ACRONYMS

A.C.	Alternating Current
AE	Architect Engineer
AEP	American Electric Power
BAST	Boric Acid Storage Tank
bcc	blind carbon copy
cc	carbon copy
CFR	Code of Federal Regulations
CHP	Chemistry Head Procedure
CR	Condition Report
D. C.	Direct Current
DCC	Donald C. Cook
DCP	Design Change Package
D/G	Diesel Generator
DRP	Division of Reactor Projects
DPR	Demonstration Power Reactor
ECCS	Emergency Core Cooling System
EDT	Eastern Daylight Time
EHP	Engineering Head Procedure
EOP	Emergency Operating Procedure
ESF	Engineered Safety Feature
FMEZ	Foreign Material Exclusion Zone
ft-lbs	Foot-Pounds
IFI	Inspection Follow Up Item
IR	Inspection Report
JO	Job Order
kV	kilo-Volts
kW	kilo-Watts
LER	Licensee Event Report
LOOP	Loss of Offsite Power
MASP	Modified Amended Security Plan
ME	Mechanical Engineering
MHP	Maintenance Head Procedure
MI	Michigan
MM	Minor Modification
MMP	Material Maintenance Head Procedure
MOD	Modification
NCV	Non-Cited Violation
NOV	Notice of Violation
NRC	Nuclear Regulatory Commission
OHP	Operations Head Procedure
OSO	Operations Standing Order
PDR	Public Document Room
PMP	Plant Manager's Procedure
PNSRC	Plant Nuclear Safety Review Committee
PSPD	Plant Set-point Document
PV	Pressure Valve
Pzr	Pressurizer

RHR	Residual Heat Removal
RMS	Radiation Monitoring System
RPP	Radiation Protection Procedure
RWST	Refueling Water Storage Tank
SE	Safety Evaluation
SI	Safety Injection
STP	Surveillance Test Procedure
S/G	Steam Generator
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
Vdc	Volts Direct Current